NON-PUBLIC?: N
ACCESSION #: 9307280187
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Calvert Cliffs, Unit 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000318

TITLE: Trip on Low Steam Generator Level Due to Insufficient Feedwater Addition
EVENT DATE: 06/25/93 LER #: 93-003-00 REPORT DATE: 07/22/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 2 POWER LEVEL: 3

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D. W. Muth, Compliance Engineer TELEPHONE: (410) 260-3592

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On Thursday, June 25, 1993 at 8:05 p.m., Calvert Cliffs Unit 2 automatically tripped on low level in the 21 Steam Generator (SG). Operators had noted divergent SG level oscillations with the Full Range Digital Feedwater Control System (FCS) in automatic. They took manual control with SG level at +25 inches but did not provide sufficient feedwater. The reactor tripped on low SG level.

The causes of this event include inadequate communication of pertinent information regarding the response of the FCS at low power, inadequate communication during shift turnover regarding a just-completed power increase, the lack of Project Team involvement at the time of the incident, and the work practices of the operations personnel involved in this incident.

Operations management will reemphasize expectations for the improvement

of communications between operating crews. An FCS Project Team representative will be available to the Control Room for future startups and planned shutdowns until FCS performance meets expectations. We will provide classroom training to all operating crews on the details of this event.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On Thursday, June 25, 1993 at 8:05:09 p.m., Calvert Cliffs Unit 2 automatically tripped on low level in the 21 Steam Generator (SG). Operators had noted divergent level oscillations with the Feedwater Control System (FCS) in automatic. They took manual control with SC level at +25 inches but did not provide sufficient feedwater. The underfeed condition lasted approximately 35 minutes at which time level reached -50 inches, tripping the reactor. The Unit was at 2.8 percent power in MODE 2 at the time of the event.

In response to previous problems controlling feedwater during startup, a new Full Range Digital FCS was installed in Unit 2 during the recently completed refueling outage. The FCS was first used during initial startup on June 12, 1993. It was noted to perform particularly well below 2 percent and above 8 percent power, with minor, gradual oscillations in SG level. However, significant oscillations were observed as power was increased between 2 and 8 percent power. These oscillations had a maximum amplitude of approximately 25 inches and period of about 30 minutes. The Operations crew performing this phase of the startup considered the oscillations acceptable when compared either with the old automatic system or with what was possible in manual and therefore did not communicate them to other Operations shifts.

On June 25, 1993 day shift had begun plant startup and had been controlling feedwater in automatic most of the day with no abnormalities noted. Power was increased from 0.1 percent to 2.8 percent just as day shift was ending. The change in power was not adequately communicated during shift turnover.

The night shift came on at about 6:00 p.m., noting the FCS in automatic and reactor power at about 3 percent. They believed that the reactor had been at this power level for some time. At about 6:30 p.m., the Control Room Supervisor (CRS) noted level oscillations beginning in the 21 SG and began monitoring level closely. At about 7:15 p.m., the CRS directed the Control Room Operator (CRO) to take manual control. The CRS had previously been informed that the feedwater regulating bypass valve had been modulating in automatic between 18 and 26 percent open throughout most of the previous shift and assumed, since he believed the plant had been at about 3 percent power during this time, that this was the proper valve setting for this power level. At the time he ordered the bypass valve to be taken into manual, the CRS noted the valve to be at 32 percent. He therefore assumed that this was too far open and that the FCS was not controlling properly.

The CRS did not realize that what had actually happened was that, with the recent increase in power to 2.8 percent, the FCS had entered the region in which increases in power produced significant SG oscillations. At the time the bypass valve was taken into manual, SG level was peaking and the FCS had closed the valve to bring level back down. With the bypass valve at 32 percent, SG 21 was actually in an underfeed condition.

The CRO took manual control with SG level still increasing and at +25 inches. He used the manual push-button controller to slightly close the valve. He noted that SG level began decreasing almost immediately and assumed that this was the result of his actions. Actually, as noted above, the FCS had already closed the valve sufficiently and the level decrease was due to this prior automatic action. The CRO began briefly depressing the control button to provide small open commands to the feedwater bypass valve with the intention of bringing level smoothly to zero. As SG level continued to decrease rapidly (3-4 inches per minute), the CRO continued to apply slight open commands to the valve that he thought more than compensated for the closure signal with which he started.

The previous SG level control system had used a knob to control valve position. The new system uses a membrane push-button with a logarithmic response. The longer the button is depressed, the faster the valve opens. The CRO was aware of the functioning of the new controller but was not aware of how little response his short presses of the button actually produced. He did not use controller output indication to obtain feedback on the effectiveness of his actions. He had previously been successful monitoring only SG level indication and controller knob position when controlling SG level with the old system.

Within 10 minutes after manual control was assumed, level dropped below zero and was not responding appreciably to operator actions with the bypass valve. The CRO continued single depressions of the controller in hopes of a gradual approach to zero inches. About five minutes later, with level approaching -15 inches, the CRO considered more aggressive feedwater injection but was concerned with the effects of level shrink if large quantities of cold water were injected into the SG. When level reached about -20 inches, the CRO took additional measures to increase level, including isolation of SG blowdown, placing the 22 feedwater bypass valve in manual and closing it slightly, and increasing 21 SG Feed Pump speed. The combination of these measures were effective in eventually terminating the drop in level at about -45 inches. However, minor level oscillations occurred, and low-level trip signals were received by Safety Channels C and D. The reactor tripped at 8:05:09 p.m. The total elapsed time for this event was approximately 45 minutes.

The appropriate Emergency Operating Procedures were performed without incident.

II. CAUSE OF EVENT

There are several causes of this event. The first is that the response of the FCS at low power was not known by Operations shifts other than the one that initially started up using the new system. This crew had observed the unanticipated oscillations, concluded that they were acceptable, and therefore did not communicate them to any other crews, The oscillations had not been discussed in training or modelled on the simulator as they had not been anticipated by the team developing the training. The crew involved in this event came on shift expecting no significant oscillations from the FCS.

A second cause is that the shift turnover did not address the increase in power that took place shortly before. This left the oncoming shift with no explanation for the SG level oscillations other than problems with the FCS. Knowledge of the power change, particularly if combined with information from the initial startup, may have helped the operators anticipate the level oscillations and either leave the FCS in automatic or understand better the need for increased feed flow in manual.

A third cause is the lack of Project Team involvement on this shift. The new FCS was sufficiently complicated that not all plant response was anticipated. A similar system at another plant maintained a flat level trend at all power levels. The response of the FCS had been compensated for during initial startup and even during the day shift prior to this event by the presence in the Control Room of a member of the FCS Project Team. This individual had proven valuable during initial startup by discussing the oscillations with the FCS manufacturer and advising the shift crew on how to respond. A member of the team could have provided a similar service during this shift.

A fourth cause of this event was the work practice of the CRO, who continued to underfeed the SG for about 35 minutes. He did not use controller output indication to verify the amount of movement the valve was making in response to his depressions of the control switch. A better understanding of the relationship of controller operation to valve movement might have resulted in more aggressive actions prior to the point where level shrinkage effects were significant. Crew supervision, including the SRO, failed to provide the CRO sufficient coaching.

We have since experienced problems with automatic FCS control causing Operators to place FCS in manual. This indicates a need to improve FCS performance to reduce challenges to SG level control.

III. ANALYSIS OF EVENT

The worst-case loss of feedwater flow transient described in the Updated Final Safety Analysis Report assumes a total loss of feedwater flow at full power and concludes that no significant safety consequences will result from this event. This analysis is bounding for this event. There are no significant safety consequences resulting from this event.

This item is reportable under the provisions of 10 CFR 50.73(A)(2)(iv) as a Reactor Protective System actuation.

IV. CORRECTIVE ACTIONS

A. Operations management will reemphasize expectations for communications between operating crews through better use of available mechanisms (i.e., operator logs, turnover sheets, and turnover briefings). Operations management will review and discuss situational leadership with Shift Supervisors.

B. An FCS Project Team representative will be available to the Control Room for future startups and planned shutdowns until FCS performance meets expectations, and Operations personnel have gained sufficient experience with the system.

C. We will provide classroom training to all operating crews on the details of this event. We will also give Operators additional hands-on training on the use of the push-button controller in conjunction with valve controller output indication for the control of steam generator level.

D. FCS will be modified to improve performance.

V. ADDITIONAL INFORMATION

A. Affected Component Identification:

IEEE 803 IEEE 805
Component or System EIIS Funct System ID

Steam Generator HX SJ
Feedwater Control System TC SJ
Feedwater Regulating Bypass Valve LCV SJ
Manual Push-button Controller LCO SJ

B. Previous Similar Events:

LER 50-317/85-009 described a trip on low SG water level from 19 percent power due to underfeeding the SG in manual.

BALTIMORE
GAS AND
ELECTRIC

CALVERT CLIFFS NUCLEAR POWER PLANT

1650 CALVERT CLIFFS PARKWAY o LUSBY, MARYLAND 20657-4702

CHARLES H. CRUSE
PLANT GENERAL MANAGER
CALVERT CLIFFS July 22, 1993

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2; Docket No. 50-318; License No. DPR 69
Licensee Event Report 93-003
Trip on Low Steam Generator Level
Due to Insufficient Feedwater Addition

The attached report is being sent to you as required under 10 CFR 50.73 guidelines. Should you have any questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

CHC/DWM/bjd
Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
R. A. Capra, NRC
D. G. McDonald, Jr., NRC
T. T. Martin, NRC
P. R. Wilson, NRC
R. I. McLean, DNR
J. H. Walter, PSC
Director, Office of Management Information
and Program Control

*** END OF DOCUMENT ***